ACCESSION #: 9606240133

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 1 PAGE: 1 OF 5

DOCKET NUMBER: 05000220

TITLE: Reactor Scram Caused by Turbine Trip Due to Feedwater

Oscillations

EVENT DATE: 05/20/96 LER #: 96-04-00 REPORT DATE: 06/19/96

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: Ken Sweet, Manager Technical TELEPHONE: (315) 349-2462

Support, Unit 1

COMPONENT FAILURE DESCRIPTION:

CAUSE: SYSTEM: COMPONENT: MANUFACTURER:

REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On May 20, 1996 at 13:18 hours, with the mode switch in "RUN", and reactor thermal power at approximately 100%, Nine Mile Point Unit 1 (NMP1) experienced a turbine trip and full reactor scram. The event was initiated by #13 feedwater flow control valve oscillations, which caused high reactor water level, which tripped the turbine and resulted in a reactor scram.

The cause of the event was a degraded actuator for the #13 feedwater flow control valve

(FCV). Upon disassembly of the actuator, worn o-rings and bushings as well as a misaligned stem and bushing were found.

Two contributing factors have been identified. The first is that the design of the current configuration of the pneumatic controls leaves little margin for degradation of the actuator and causes the valve to operate at a high frequency cycle. This cycling may cause increased wear on the actuator bushings and rubber goods which could lead to volume booster needle valve clogging and cause instability in the control system.

The second is the misalignment of the actuator stem. This misalignment caused binding which could also have caused erratic operation. The probable cause of the misalignment was incorrect factory assembly of the actuator, resulting in a misalignment of the internal bushings.

The immediate corrective actions included commencing scram recovery activities and initiating a controlled plant cooldown. Other corrective actions included: replacing the valve positioner volume boosters and rebuilding the actuator. In addition, the control system design is being evaluated to determine appropriate measures to increase reliability.

TEXT PAGE 2 OF 5

I. DESCRIPTION OF EVENT

On May 20, 1996 at 1318 hours, NMP1 experienced a turbine trip and full reactor scram. Prior to the trip, the plant was operating at 100 percent power. The immediate cause of the transient was high reactor water level which subsequently tripped the turbine and resulted in a reactor scram. No related testing or plant evolutions were in progress at the time of the scram.

The cause of the high water level was failure of #13 feedwater flow control valve (FCV). Operations had been alerted a few minutes before the scram that #13 FCV was oscillating. Operators took manual control of the flow control valve (from the control room) but feedwater flow oscillations continued. The #12 feedwater pump was started, and reactor

power was being reduced via recirculation flow to attempt to secure #13 feedwater pump. Locally, operators were attempting to take local manual control of the valve but the valve movement made it difficult to align the forked pin and latch the valve to the manual hand wheel mechanism.
Following the scram, powerboards 11 and 12 fast transferred as designed, and no significant electrical transient occurred. Reactor water level decreased to +34 inches, and Emergency Operating Procedure N1-EOP-2, "RPV Control (EOP-2)" was entered. The High Pressure Coolant Injection System (HPCI) initiated and increased the level to normal as designed.
All control rods inserted to 00 during the scram as verified by the process computer. The remote shutdown panel indication did not show an all-rods-in light lit as would be expected. Evaluation determined that rod 34-27 did not have either of the two "green" reed switches made up which feed the all-rods-in light. This caused the all-rods-in light to remain unlit. Indications showed that this rod was at a higher temperature than others. Therefore, it is believed that the effects described in SIL 532 were observed. SIL 532 describes problems with all-rods-in (ARI) indication due to thermal effects on the magnet assemblies in the position indicators. The remote shutdown panels have been declared operable per N1-ODP-OPS-0103.
Individual control rod 90 percent scram insertion times for 8 selected control rods were evaluated as required by Tech. Spec. 3/4.1.1 and found to be satisfactory. All 30 rods (90 percent scram insertion times)

recorded on the 30 channel recorder were satisfactory.

FSAR transient analysis for a scram from 100 percent power shows peak reactor pressure at 1116 psig and all 6 ERVs opening. This transient was initiated at a lower reactor power level (approximately 80 %), therefore, no ERVs opened. Peak reactor pressure was 1075 psig as recorded by Data Acquisition Analysis System (DAAS) and 1070 psig via the control room chart recorder. The high reactor pressure scram RPS signal annunciated as expected.

II. CAUSE OF THE EVENT

A cause analysis of the event was performed utilizing Nuclear Interfacing Procedure NIP-ECA-01, "Deviation/Event Report." The cause of the event was determined to be a degraded valve actuator. Two contributing factors have been identified; pneumatic control system design and misalignment of the valve and actuator stem. The original design of the pneumatic control system left little margin for pneumatic control component degradation. The initial design of the 13 FCV in the 1993 refuel outage 12 (RFO12) left the minimum booster relay needle valve position only 1/8 turn open minimizing air flow. This left the valve susceptible to actuator wear particles caused by the continuous control action. Wear particles may have migrated from the actuator assembly and become lodged in the booster relay needle valves. This would have caused the feedwater control valve to have a higher gain, causing flow control to become unstable.

The second contributing factor is the misalignment of the actuator stem. This misalignment caused binding which could also have caused erratic operation. During the rebuild of the actuator, the stem and stem bushings were found misaligned. The probable cause of the misalignment was incorrect factory assembly of the actuator resulting in a misalignment of the internal bushings.

III. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR50.73(a)(2)(iv), which requires licensees to report "any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

The turbine trip and resultant reactor scram was the design response of the RPS to high reactor water level. The HPCI mode of the feedwater system initiated to maintain reactor water level as designed.

An evaluation of the effect of the transient on the core shroud and supports was performed by Design Engineering and this transient did not induce any unacceptable stresses. Also the effects of this feedwater transient on the feedwater inlet nozzles were evaluated, and no concerns were identified.

There were no adverse safety consequences as a result of the event. No systems or components were inoperable that contributed to the severity of the event. The reactor scram posed no threat to the health and safety of the general public or plant personnel.

IV. CORRECTIVE ACTIONS

Immediate Corrective Actions:

1. Operators performed the scram recovery actions and placed the plant in a stable condition.

2. Technical Support, Engineering, and Maintenance personnel performed detailed troubleshooting of the valve and actuator to determine the cause of failure.

3. The valve actuator was rebuilt, the positioner and booster relay valves were replaced, and the FCV was successfully tested and returned to service.

Additional corrective actions include:

1. An evaluation of the preventative maintenance process was performed and verified that vendor recommendations had previously been met for this actuator and valve.

2. Applicable maintenance procedures will be developed and/or revised to ensure proper alignment of actuator and valve stems. The preventative maintenance frequency for the valve actuator will be changed from the vendor recommended five years to two years. This revision will insure proper actuator alignment and replace the actuator o-ring and the bushings. This will be complete by December 15, 1996.

3. The control system design will be re-evaluated to determine appropriate measures to increase reliability. This may include recommendations for modifications to increase pneumatic volume of subcomponents to decrease the gain of the valve controls and limit cycling of the feedwater control valve. This will allow the booster to be less susceptible to wear material and reduce the number of service cycles on the boosters. This will be complete by October 31, 1996.

V. ADDITIONAL INFORMATION

A. Failed Components:

Valve actuator; Control Components, Inc., 313 in**2.

B. Previous similar events: None.

V. ADDITIONAL INFORMATION (cont'd)

C. Identification of components referred to in this LER:

COMPONENT IEEE 803 EIIS FUNCTION IEEE 805 SYSTEM ID

Reactor Protection System N/A JC

Main Turbine/Generator N/A TA

Main Feedwater System N/A SJ

High Pressure Coolant

Injection System N/A BJ

Reactor Pressure Vessel N/A SB

Control Rod ROD SB

Flow Control Valve FCV SJ

ATTACHMENT 1 TO 9606240133 PAGE 1 OF 1ATTACHMENT 1 TO 9606240133 PAGE 1 OF 1

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RICHARD B. ABBOTT

Vice President and June 19, 1996

General Manager - Nuclear NMP1L 1085

U. S. Nuclear Regulatory Commission

Attn: Document Control Desk

Washington, DC 20555

RE: Docket No. 50-220

LER 96-04

Gentlemen:

In accordance with 10CFR50.73 (a)(2)(iv), we are submitting LER 96-04,

"Reactor Scram Caused by Turbine Trip Due to Feedwater Oscillations."

A telephone report of this event was made in accordance with 10CFR50.72

(b)(2)(ii) at 1413 hours on May 20, 1996.

Very truly yours,

R. B. Abbott

Vice President and General Manager - Nuclear

RBA/TWR/lmc

Attachment

xc: Mr. Thomas T. Martin, Regional Administrator, Region I

Mr. Barry S. Norris, Senior Resident Inspector

*** END OF DOCUMENT ***